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# A Study on Spatial Neutron Kinetics of a Pressurized Water Reactor

Chang Hyo Kim
Seoul National University
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# 가압경수로의 공간의존적 핵적동특성에 관한 연구

김 창 효
서울대학교
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## Abstract

The purpose of this work is to present a spatial neutron kinetics computational scheme for the analysis of space-dependent transients like rod ejection accident of pressurized water reactors. In this work modified Borresen's 1.5 group coarse mesh scheme was formulated for the neutronic computation of the space-dependent transients and applied to the analysis of hypothetical rod ejection accident of KNU no. 1 PWR core at BOC, HZP. The computational accuracy of the modified Borresen's scheme is examined by comparing calculations for core power and control rod worths with startup core physics test results. Effects of such parameters as ejected rod worths and number of delayed neution group on transient results as well as computational efficiency are also examined. On this basis it is suggested that the modified Borresen's method is a useful scheme for the analysis of spatial neutron transients of PWR's.

## 요 약

본 논문은 가압 경수형 원자로의 제어봉 이탈사고와 같이 공간 의존적 과도특성 해석에 필히 요구되는 가상적 사고 분석을 위한 핵적 동특성 코드의 개발을 위한 것이다. 본 논문에서는 1.5군 중성자 확산 방정식에 의거한 수정형 Borresen 모형을 핵적 동특성 모델로 잡고 이를 공간의존적 과도특성해석에 응용할 수 있도록 수식화 하여 고리 1호기 초기 노심의 가상적인 제어봉 이탈 사고 해석에 응용했다. 본 사고 해석에 채택한 수정형 Borresen 모형에 대한 계산 정밀도의 검증을 위해 출력 분포 및 제어봉가등 계산결과를 고리 1호기 초기 노심의 노물리 실험자료와 비교 했고 공간의존적 사고해석에 있어서 중시되는 핵적 동특성 방정식의 계산 효율성을 검토했다. 그리고 이 결과를 토대로 수정형 Borresen 모형이 제어봉 이탈사고, 증기관 파탄사고 등과 같은 공간의존적 사고 해석에 유용하게 이용될 수 있다는 것을 보였다.

## 1. Introduction

The reactor transients such as control rod ejection and steam line break accidents are associated with local changes in the core composition and properties. In the loosely coupled LWR cores, these changes in the material properties result in a striking power tilting around the immediate vicinity of the affected core region. For the analysis of such accidents, therefore, the space-dependent kinetics computation is required to predict accurately the time behavior of the core power during the transient period. The purpose of this paper is to present a mordified Borresen's 1.5 group diffusion theory method[1,2] as a computational scheme for the spatial neutron kinetics analysis.

One of important concerns in the space-dependent kinetics analysis is how to reduce the long computing time required for obtaining the multi-dimensional solution to the group diffusion equations.[3] The finite difference method as represented by the MEKIN code can be adopted,[4] yet it has a major drawback of prohibitedly long computing time, especially in 3-D computation. The modern coarse-mesh methods are designed to get around this drawback.[5,6] The modified Borresen's method adopted in this papar is one of the simplest among various coarse-mesh methods. Unlike most of coarse-mesh methods, the method is based on the 1.5 group scheme. The method was originally suggested to improve the computational accuracy of the Borresen's method[7] in predicting the steady-state power distribution of the LWR. It has been shown recently that the method is also applicable to the spatial neutron kinetics by the LRA-BWR benchmark problem.[2] Since the method has not been fully tested in terms of real reactor problems, we intend here to make a further investigation on the effectiveness of the method in the applications to the space-dependent kinetics problem.

The problem to be investigated is the hypothetical rod withdrawal accident of the KNU No. 1 PWR core. This accident results from the mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The accident corresponds to fast reactivity transients resulting in rapid power excursion and pronounced power peaking in the immediate vicinity of the ejected control rod. The transient is initally suppressed by the fast-acting Doppler feedback effects and eventually by reactor scram triggered by the high neutron flux set point. For this study, the rod withdrawal accident is treated as ATWS. It is investigated whether or not the power excursion is safely suppressed by the Doppler effect.

## 2. Description of the Method

The modified Borresen's method for the kinetics applications is described in detail in reference 2. Therefore, some of the mathematical details in describing the method will be referred to the reference 2 in the below.

The equations to be solved in the modified Borresen's method are the nodal balance relations for the fast neutrons,

$$\frac{1}{v_f}\frac{d}{dt}\bar{\phi}_{fm}(t)=-\frac{1}{V_m}\int_{Vm}\vec{J}_f(t)\cdot ds -\Sigma_{tfm}\bar{\phi}_{fm}(t)+\frac{1}{k_{eff}}(1-\beta)(\nu\Sigma_{ffm}\bar{\phi}_{fm}(t)+\nu\Sigma_{ftm}\bar{\phi}_{tm}(t))+\sum_i\lambda_i\bar{C}_{im}(t). \qquad (1)$$

and the delayed neutron precursors

$$\frac{d}{dt}\bar{C}_{im}(t)=\frac{\beta_i}{k_{eff}}(\nu\Sigma_{ffm}\bar{\phi}_{fm}(t)+\nu\Sigma_{ftm}\bar{\phi}_{tm}(t))-\lambda_i\bar{C}_{im}(t)\ ;\quad i=1,2,\cdots 6. \qquad (2)$$

The $\bar{\phi}_{fm}(t)$ is the fast group flux averaged over a nodal volume $m$ at time $t$, and $J_f(t)$ is fast neutron current at the surfaces of node $m$ at time $t$. The $\bar{C}_{im}(t)$ is the density of delayed neutron precursor group $i$ averaged over the nodal volume.

Discretizing Eqs. (1) and (2) using the Borresen's principle for the spatial differencing[4] and the fully implicit approximation for the temporal differencing, one finds a coarse-mesh difference relations for the diffusion density,

$$\varphi_m^{(n)} = \sqrt{D_{fm}}\cdot\phi_{fm}^{(n)},$$

$$-\sum_{4j}\varphi_j^{(n)} - R\cdot\sum_{2j}\varphi_j^{(n)} + Q_m^{(n)}\,\varphi_m^{(n)} = \frac{S_m^{(n)}}{h_x^2(1-c_f q_m^{(n)})\sqrt{D_{fm}}} \quad (3)$$

where

$$S_m^{(n)} = \frac{1}{v_f\Delta t_n}\,\bar{\phi}_{fm}^{(n-1)} + \sum_i \frac{\lambda_i}{1+\lambda_i\Delta t_n}\,\bar{C}_{im}^{(n-1)} + \frac{1}{k_{eff}}\left(1-\sum_i\frac{\beta_i}{1+\lambda_i\Delta t_n}\right)\Sigma_{ftm}\,\bar{\phi}_{tm}^{(n)}, \quad (4a)$$

$$Q_m^{(n)} = \frac{p_m+(b_f+c_f r_m)\cdot q_m^{(n)}}{1-c_f q_m^{(n)}}, \quad (4b)$$

$$P_m = \sum_{4j}\sqrt{\frac{D_{fj}}{D_{fm}}} + R\cdot\sum_{2j}\sqrt{\frac{D_{fj}}{D_{fm}}}, \quad (4c)$$

$$\gamma_m = \sum_{4j}\sqrt{\frac{D_{fm}}{D_{fj}}} + R\cdot\sum_{2j}\sqrt{\frac{D_{fm}}{D_{fj}}}, \quad (4d)$$

$$q_m^{(n)} = \frac{h_x^2}{D_{fm}}\left\{\frac{1}{v_f\Delta t_n} + \Sigma_{afm}^{(n)} + \Sigma_{rm}^{(n)} - \frac{1}{k_{eff}}\left(1-\sum_i\frac{\beta_i}{1+\lambda_i\Delta t_n}\right)\cdot\nu\Sigma_{ffm}\right\}. \quad (4e)$$

The right hand side of Eq. (3), the source term which drives the diffusion density at time step $t_n$, $\varphi_m^{(n)} = \sqrt{D_{fm}}\cdot\phi_{fm}^{(n)}$, contains the thermal group flux $\bar{\phi}_{tm}^{(n)}$. In the modified Borresen's method, this is approximated by an interpolation formula,

$$\bar{\phi}_{tm}(t) = b_{tm}(t)\phi_{tm}(t) + 2c_{tm}(t)\cdot\left\{\sum_{4j}\phi_{tm}^j(t) + R\sum_{2j}\phi_{tm}^j(t)\right\} \quad (5)$$

in combination with the analytic expression for thermal group flux at the internodal surface $\phi_{tm}^{j(n)}$.

$$\phi_{tm}^j(t) = \frac{\dfrac{D_{tm}k_m(t)}{T_m(t)}\phi_{tm}(t) + \dfrac{D_{tj}k_j(t)}{T_j(t)}\phi_{tj}(t)}{\dfrac{D_{tm}k_m(t)}{T_m(t)} + \dfrac{D_{tj}k_j(t)}{T_j}} \quad (6)$$

where

$$k_m(t) = \sqrt{\left(\Sigma_{atm}(t) + \frac{w_{tm}}{v_t}\right)/D_{tm}}, \quad (7a)$$

$$T_m(t) = t\tanh(k_m(t)h_u/2)\ ;\ u=x \text{ or } z. \quad (7b)$$

The $\phi_{tm}^{(n)}$, thermal flux at the center of the node $m$, is obtained in analogy with the static case by

$$\phi_{tm}(t) = \frac{\Sigma_{rm}\,\bar{\phi}_{fm}(t)}{\dfrac{w_m}{v_t} + \Sigma_{atm}(t)} \quad (8)$$

As for the thermal weight factor $b_m(t)$, an argument similar to the static case[1]. leads to

$$b_{tm}(t) = (1-2\tanh(k_m(t)h_x/2)/k_m(t)h_x)^2\cdot(1-2\tanh(k_m(t)h_z/2)/k_m(t)h_z) \quad (9)$$

The $c_{tm}(t)$ is then determined by the relation,

$$b_{tm}(t) + 2c_{tm}(t)(4+2R) = 1. \quad (10)$$

## 3. Results and Discussion

The modified Borresen's formulation given above is applied to the analysis of the hypothetical control rod ejection accident of the KNU No. 1 PWR. The reactor core contains 121 fuel assemblies in three different enrichent regions. Fig. 1 displays the three-region fuel loading

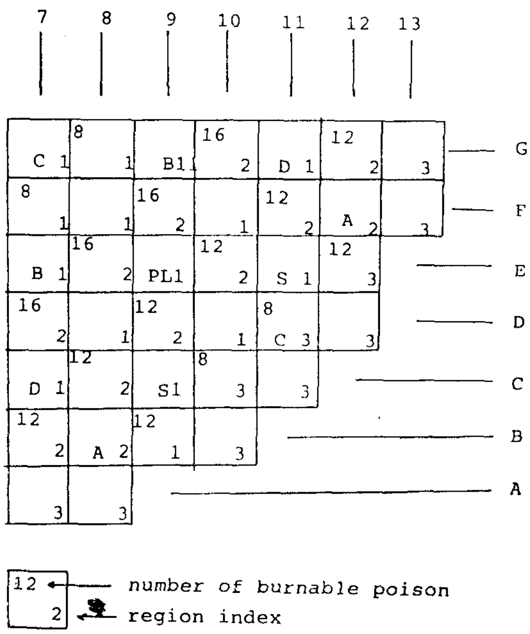


**Fig. 1. A Quarter Core of KNU No. 1 PWR.**

**Table 1. Input Data for the Analysis of Rod Ejection Accident**

A. Cross section model

Fast neution absorption cross section

$$\Sigma_{a1}=\Sigma_{a1,0}\{1+\alpha(\sqrt{T_{eff}}-\sqrt{T_0})\}+(\rho-\rho_0)\sigma^{w}_{a1}+\Delta\Sigma^{CR}_{1}$$

Thermal neution absorption cross section

$$\Sigma_{a2}=\left(1+0.12\frac{\rho-\rho_0}{\rho_0}\right)\{\Sigma_{a2,0}+(\rho-\rho_0)\sigma^{w}_{a2}\}+\Delta\Sigma^{CR}_{2}$$

Removal cross section

$$\Sigma_r=\Sigma_{a1}/(e^{\Sigma_{a1}/\Sigma_m}-1)\,;\ \Sigma_m=\Sigma_{m,0}+(\rho-\rho_0)\sigma^{w}_{m}$$

B. Two-group constants of fuel material, $\Sigma_{x,0}$

| Fuel Type* | | | $D_g$ | $\Sigma_{ag}$ | $\Sigma_R$ | $\nu\Sigma_{fg}$ | $\kappa\Sigma_{fg}$ |
|---|---|---|---|---|---|---|---|
| ID | Reg. | NBP | | | | | |
| 1 | 1 | 0 | 1.3436E+00 | 9.1435E−03 | 1.8525E−02 | 5.6438E−03 | 7.6387E−14 |
| | | | 4.5131E−01 | 1.4453E−01 | 0.0000E+00 | 1.7976E−01 | 2.3951E−12 |
| 2 | 1 | 8 | 1.3490E+00 | 9.5326E−03 | 1.7660E−02 | 5.6456E−03 | 7.6411E−14 |
| | | | 4.5553E−01 | 1.6061E−01 | 0.0000E+00 | 1.8172E−01 | 2.4213E−12 |
| 3 | 2 | 0 | 1.3455E+00 | 9.6834E−03 | 1.8168E−02 | 6.6613E−03 | 8.9161E−14 |
| | | | 4.4161E−01 | 1.7157E−01 | 0.0000E+00 | 2.3527E−01 | 3.1237E−12 |
| 4 | 2 | 12 | 1.3535E+00 | 1.0275E−02 | 1.6854E−02 | 6.6681E−03 | 8.9252E−14 |
| | | | 4.4779E−01 | 1.9671E−01 | 0.0000E+00 | 2.3866E−01 | 3.1687E−12 |
| 5 | 2 | 16 | 1.3562E+00 | 1.0469E−02 | 1.6423E−02 | 6.6686E−03 | 8.9259E−14 |
| | | | 4.4993E−01 | 2.0505E−01 | 0.0000E+00 | 2.4019E−01 | 3.1891E−12 |
| 6 | 3 | 0 | 1.3462E+00 | 9.9544E−03 | 1.7979E−02 | 7.1642E−03 | 9.5807E−14 |
| | | | 4.3693E−01 | 1.8493E−01 | 0.0000E+00 | 2.6270E−01 | 3.4843E−12 |
| 7 | 3 | 8 | 1.3516E+00 | 1.0350E−02 | 1.7103E−02 | 7.1695E−03 | 9.5878E−14 |
| | | | 4.4099E−01 | 2.0149E−01 | 0.0000E+00 | 2.6551E−01 | 3.5215E−12 |
| 8 | 3 | 12 | 1.3543E+00 | 1.0546E−02 | 1.6666E−02 | 7.1714E−03 | 9.5903E−14 |
| | | | 4.4306E−01 | 2.1031E−01 | 0.0000E+00 | 2.6656E−01 | 3.5354E−12 |

C. Doppler coeffieients of fuel material, $\alpha(\times 10^5)$

| Fuel Type, ID | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 |
|---|---|---|---|---|---|---|---|---|
| $\alpha$ | 3.0228 | 3.0231 | 3.0137 | 3.0165 | 3.0165 | 3.0162 | 3.0165 | 3.0190 |

D. Incremental cross sections of control rod, $\Delta\Sigma^{CR}_{g}$ (g=1, 2)

| Fuel Type, Reg | $\Delta\Sigma^{CR}_{1}$ | $\Delta\Sigma^{CR}_{2}$ |
|---|---|---|
| 1 | $.33416\times10^{-2}$ | .05345 |
| 2 | $.38125\times10^{-2}$ | .06100 |

E. miscellaneous cross section data.

| Fuel Type | | | $\sigma^{a_1}{}_{w}$ | $\sigma^{a_2}{}_{w}$ | $\sigma^{m}{}_{w}$ |
|---|---|---|---|---|---|
| ID | Reg | NBP | | | |
| 1 | 1 | 0 | 2.3619E−3 | 2.5071E−02 | 2.8390E−02 |
| 2 | 1 | 8 | 2.3214E−3 | 1.8241E−02 | 2.7517E−02 |

| | | | | | |
|---|---|---|---|---|---|
| 3 | 2 | 0 | 2.4663E−3 | 1.9570E−02 | 2.8249E−02 |
| 4 | 2 | 12 | 2.4215E−3 | 1.0655E−02 | 2.6886E−02 |
| 5 | 2 | 16 | 2.4033E−3 | 8.4148E−03 | 2.6460E−02 |
| 6 | 3 | 0 | 2.5330E−3 | 1.6344E−02 | 2.8145E−02 |
| 7 | 3 | 8 | 2.5114E−3 | 1.0343E−02 | 2.7268E−02 |
| 8 | 3 | 12 | 2.4846E−3 | 7.7565E−03 | 2.6810E−02 |

F. Delayed neutron data

| Delayed $n$ groups | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| $\beta_i$ (%) | .0127 | .1463 | .1352 | .2825 | .0963 | .0323 |
| $\lambda_i$ (sec$^{-1}$) | .0125 | .0308 | .1155 | .3116 | 1.2479 | 3.3531 |
| $\beta$ (%) | .7143 | | | | | |
| $\lambda$ (sec$^{-1}$) | .0825 | | | | | |

* ID : identification index Reg: region index
NBP: number of burnable poison rods

pattern together with rod cluster control assembly arrangement. The rod ejection accident of this reactor is assumed to take place by the accidental ejection of the *C* bank control assembly at *D*11 at the speed of 36.75m/sec. The reactor is initially in the hot standby condition at the BOL with control bank *C* and *D* fully inserted.

The input data required for the analysis of this accident include group cross sections of different fuel meterial in which the reactivity feedback effects are adequately accounted for. Table 1 summarizes the cross section model and related numerical data. It is noted that the feedback effects are described by representing the absorption and removal cross section as functions of fuel temperature and moderator density. The Doppler effect is described by the fast group absorption cross section which is given by a function of square root of fuel temperature.

The computation of the feedback effects requires thermal hydraulic model for the core heat transport. For this study, a singly lumped fuel temperature model[8] is adopted.

$$M_f C_f \frac{dT_{fm}}{dt} = P_{fm} - \frac{1}{R}(T_{fm} - T_o).$$

The cross section inputs of fuel assemblies in Table 1 are obtained from homogenizat ion computations of individual fuel assemblies using the lattice cell code KICC[9] in combination with the 2-dimensional assembly homogenization code, KIDD[10]. The incremental absorption cross sections of control rods, $\Delta\sum_g^{CR}(g=1,\ 2)$, are obtained from the ARMP formula.[11] The transient results are strongly dependent on these data. Therefore, the validity of these data is examined in terms of the core power distribution at HZP, BOL and reactivity worths of control banks. Figs. 2 and 3 show that computations for core power distribution agree well with the startup core physics test at BOL.[12] The mean relative errors in the normalized assembly power densities are within 2.5% for both 2-D and 3-D computations. Table 2 shows that the 2-D

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| 1.049<br>1.064<br>1.042<br>1.040 | .960<br>.977<br>.973<br>.969 | 1.097<br>1.107<br>1.080<br>1.095 | 1.111<br>1.125<br>1.124<br>1.130 | 1.147<br>1.122<br>1.116<br>1.151 | 1.076<br>1.056<br>1.063<br>1.083 | 0.806<br>.829<br>.838<br>.846 | measurement<br>3-D, 1N/FA<br>2-D, 1N/FA<br>2-D, 4N/FA |
| | 1.081<br>1.091<br>1.072<br>1.072 | 1.083<br>1.116<br>1.111<br>1.104 | 1.158<br>1.148<br>1.130<br>1.147 | 1.185<br>1.161<br>1.162<br>1.172 | 1.171<br>1.162<br>1.149<br>1.133 | 0.658<br>.676<br>.689<br>.664 | |
| | | 1.157<br>1.142<br>1.129<br>1.146 | 1.202<br>1.178<br>1.134<br>1.193 | 1.069<br>1.010<br>1.056<br>1.043 | 0.770<br>.760<br>.779<br>.765 | | |
| | | | 1.149<br>1.082<br>1.081<br>1.105 | 1.052<br>1.056<br>1.056<br>1.042 | .591<br>.613<br>.628<br>.589 | | |
| | | | | .711<br>.710<br>.722<br>.689 | | | |

**Fig. 2. Normalized Fuel Assembly Power Distribution of KNU No. 1 PWR at BOC, HZP ARO. 1N/FA=1 node per fuel assembly 4N/FA=4 nodes per fuel assembly**

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| .718<br>.641<br>.666<br>.684 | 1.047<br>1.020<br>1.016<br>1.036 | 1.288<br>1.247<br>1.225<br>1.270 | 1.094<br>1.114<br>1.109<br>1.130 | .678<br>.634<br>.668<br>.690 | 1.024<br>1.038<br>1.048<br>1.061 | .894<br>.947<br>.954<br>.971 | measurements<br>3-D, 1N/FA<br>2-D, 1N/FA<br>2-D, 4N/FA |
| | 1.243<br>1.246<br>1.223<br>1.236 | 1.237<br>1.278<br>1.271<br>1.280 | 1.227<br>1.207<br>1.191<br>1.212 | 1.133<br>1.119<br>1.115<br>1.120 | 1.214<br>1.259<br>1.240<br>1.198 | .725<br>.787<br>.800<br>.767 | |
| | | 1.271<br>1.253<br>1.235<br>1.249 | 1.154<br>1.138<br>1.121<br>1.145 | 1.084<br>1.046<br>1.015<br>1.042 | .868<br>.823<br>.841<br>.821 | | |
| | | | .641<br>.580<br>.612<br>.624 | .929<br>.928<br>.925<br>.910 | .639<br>.623<br>.637<br>.604 | | |
| | | | | .620<br>.658<br>.667<br>.614 | | | |

**Fig. 3. Normalized Fuel Assembly Power Distribution of KNU No. 1 PWR at BOC, HZP, D and C in.**

computations for rod worths agree with measurements within 2% but that the 3-D computations underestimates the rod worths by 10%. The apparently less accurate rod worth estimation by the 3-D method than the 2-D method is a fortuitous outcome resulting from the use of the incremental cross sections by the ARMP formula, an approximate empirical formula. Therefore it must not be understood that the 3-D method is less accurate than the 2-D method.

The hypothetical ejection of control bank C with the speed of 36.75m/s corresponds to the accidental reactivity insertion of more than 1 $ in the time period of 0.1 second. Fig. 4 shows the 2-and the 3-dimensional modified Borresen's model computations for the core power response to the accidental reactivity insertion. The figure displays a characteristic core power behavior accompaning the rod ejection accident, e.g., a power burst in which the core power rises rapidly right after the accident, reduces subsequently due to the Doppler effect, and stabilizes to the steady-state value. Table 3 presents summary of the 2-D and the 3-D computations for the transient core power behavior in the time period of 3 seconds. Two computations predict the timing of the power peak and duration of the power burst almost identically, yet

**Table 2. Control Rod Worths at Bol, HZP, No Xenon** (PCM)

| Rod Configuration | Measurements (cumulative) | 2-D | | 3-D |
|---|---|---|---|---|
| | | 1N/FA | 4N/FA | |
| D | 973 | 960 | 987 | 916 |
| D+C | 2473 | 2327 | 2409 | 2217 |
| D+C+B | 3443 | 3280 | 3380 | 3146 |
| D+C+B+A | 5339 | 5363 | 5353 | 5078 |
| D+C+B+A+S | 7615 | 7225 | 7371 | 6938 |
| Ejected Rodds | | | | |
| C rods at G 7 | — | 363 | 365 | 339 |
| C rods at D11 | — | 1180 | 1167 | 1066 |
| D rods at G11 | — | 863 | 797 | 742 |

**Table 3. Summary of Results for 2-D and 3-D Rod Ejection Accident Analysis**

| Parameter | 2-D | 3-D |
|---|---|---|
| initial core state | HZP | HZP |
| (fraction of noninal core thermal power) | $(10^{-6})$ | $(10^{-6})$ |
| $k_{eff}$ for the initial state | .998884 | .995120 |
| fuel/moderator temperature, °F | 547 | 547 |
| reactivity of ejected rod, $ | 1.63 | 1.49 |
| rod ejection velocity, m/sec | 36.75 | 36.75 |
| delayed neutron groups | 1 | 1 |
| time to power peak, sec | .166 | .169 |
| power density at the peak, w/cc | 15833 | 7366 |
| peak/average fuel temperature at 3 sec, °F | 1224/1061 | 1085/872 |
| Computer System | MV-8000 | MV-8000 |
| CUP time, sec. | 142 | 4534 |

the 2-D prediction on the peak power density is much higher than the 3-D prediction. This difference of the peak power density is due to underprediction of ejected rod worths in 3-D computation, as shown in Table 2. The peak power density given in Table 3 is 1-or 2-order higher than the core average power density at HFP, 98.8 w/cc. However, the duration of the power burst is so short that the fuel temperatare increases but remains below the nominal full power temperature. Fig. 5 reflects this behavior for fuel temperature.

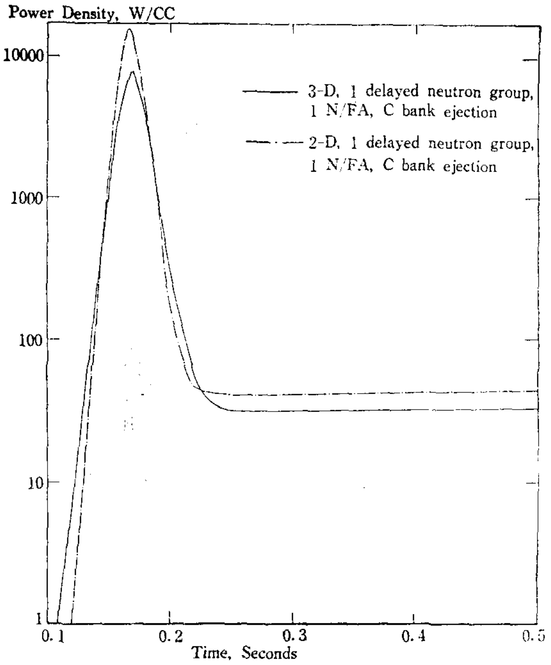


**Fig. 4. Mean Core Power Density (*W*/cc) after Ejection of C Bank at D11**

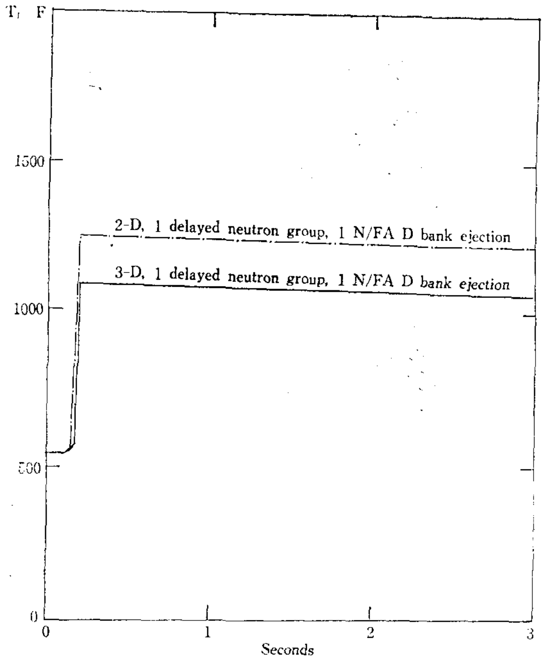


**Fig. 5. Fuel Temperature Behavior after C Bank Ejection at HZP**

Transient power behavior shown in Fig. 4 may be affected by the number of delayed neutron groups, and the ejected rod worths. Fig. 6 displays effects of these parameters on the transient results. Comparison of curves a and b show that the power burst predicted by using 6 delayed neutron groups stabilizes earlier with higher power density than the one by one delayed neutron group model. This is because the short-lived delayed neutron precursors in the six group scheme contributes to the neutron population earlier than the one group scheme.

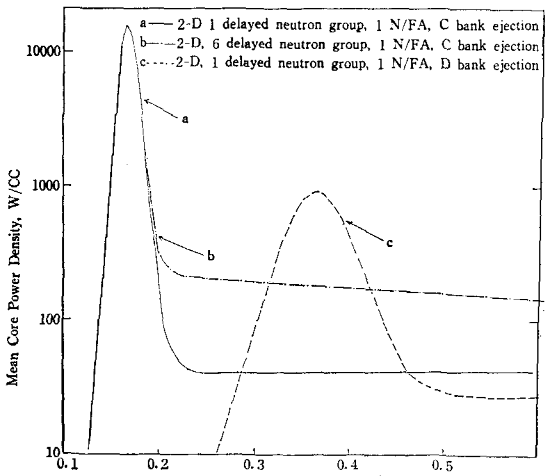


**Fig. 6. Effects of the Number of Delayed Neutron Group and Ejected Rod Worth on Transient Power.**

The reactivity worth of ejected rod has pronounced effect on the transient power behavior. As is shown in curve c of Fig. 6, the ejection of D bank rod at G11 leads to power burst in which the timing of the peak is much delayed and the power peak is one order lese than, the case of C bank rod ejection. It is noted that the D and E bank worths are 1.63$ and 1.49$, respectively.

## 4. Conclusion

Three-dimensional computation is required for the accurate prediction of the space-and time-dependent effects in large PWR cores with feedback, yet the computing cost has been a major concern in the 3-D calculation. In this respect the computing efficiency of the modified Borresen's method as shown in Table 3 is quite remarkble. The computing time of about 4,500 seconds involues a total of 2560 temporal steps on the time step sizes of 0.001 s till 0.2 seconds and 0.005 s for 0.2 to 3 seconds and is tolerable enough to justify routine applications of the modified Borresen's scheme for the 3-D spatial neutron transient analysis of PWR's.

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